NON-PUBLIC?: N
ACCESSION #: 9001250197
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Peach Bottom Atomic Power Station - PAGE: 1 OF 06
Unit 2

DOCKET NUMBER: 05000277

TITLE: Malfunctioning Electro-Hydraulic Control System Component Causes Scram When Removed From Service
EVENT DATE: 07/21/89 LER #: 89-015-01 REPORT DATE: 01/16/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 079

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: T. E. Cribbe, Regulatory Engineer TELEPHONE: (717) 456-7014

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: TG COMPONENT: PE MANUFACTURER: G084
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 2231 on 7/21/89 with Unit 2 at 79% thermal power, an attempt was made to remove a malfunctioning Reactor Pressure Vessel (RPV) Pressure Regulator Set from the electronic portion of the Main Turbine (MT) Electro-Hydraulic Control (EHC) Pressure Regulating System. Immediately, the MT Bypass and Control Valves opened, causing main steam line pressure to decrease to approximately 480 psig. At 850 psig main steam line pressure a Group I Isolation occurred causing the Main Steam Isolation Valves (MSIV) to close. As a result, a full reactor scram occurred. RPV level decrease due to shrink following MSIV closure resulted in a Group II and III isolation as level decreased below 0 inches. Two Main Steam Relief Valves (MSRV) lifted once automatically, followed by manual Operator cycling of MSRVs to control RPV pressure between 930 psig and 1060 psig. The High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems were placed in operation to control RPV

pressure and level. The root cause of this event was a malfunction of the electronic portion of the "A" RPV Pressure Regulator Set. No actual safety consequences occurred as a result of this event. The majority of the "A" Regulator electronic components were replaced. This event has been reviewed with appropriate plant personnel. One previous similar LER was identified.

END OF ABSTRACT

TEXT PAGE 2 OF 06

Requirements for the Report

This LER is being submitted pursuant to 10CFR50.73(a)(2)(iv) to report those conditions which resulted in the automatic actuation of an Engineered Safety Feature i.e., Reactor Protection System (EIIS:JC) and Primary Containment Isolation System (EIIS:JM).

Unit Status at Time of Event

Unit 2 was in the Run Mode at 79% thermal power increasing generator (EIIS:TG) load at a rate of 10 MWe per hour. Unit 3 was in the Refuel Mode with the core offloaded. Unit 2 "A" RPV Pressure Regulator (EIIS:RG) Set output signal was drifting slowly upward at a constant rate (pressure decreasing).

Description of Event

NOTE: The Main Turbine (MT) (EIIS:TRB) Electro-Hydraulic Control (EHC) (EIIS:TG) Pressure Regulating System consists of two redundant Pressure Regulator (EIIS:RG) Sets which maintain constant reactor (EIIS:RCT) pressure in coordination with MT speed and load control.

On 7/3/89 while performing MT EHC Pressure Regulating System stability testing (with the "A" Pressure Regulator in service) it was noted that the signal output from "B" Pressure Regulator circuit was offset from the "A" approximately 15 psig (normal 3 psig). The offset was returned to the 3 psig setting by adjusting the pressure setpoint bias potentiometer. Continued observation revealed the pressure differential between regulator circuits to be increasing requiring periodic bias potentiometer adjustments to maintain 3 psid between channels (EIIS:CHA). Testing indicated the "A" Regulator signal output was increasing slowly (causing pressure to decrease 0.4 psig/day), while at the same time "B" Regulator signal output appeared stable. It was noted that the bias potentiometer was quickly running out of adjustment requiring action before a loss of the ability to maintain the required 3 psid between regulators occurred.

The 3 psid value is required to enable a relatively smooth transfer to the redundant regulator in the event of failure of the regulator in service. On 7/20/89 the Plant Manager was informed of the problem and options available.

On 7/21/89 a meeting attended by the Plant Manager, System Engineering, Shift Supervision and a General Electric (GE) representative with a knowledge of EHC was held to evaluate each available option and its associated risk in order to determine an action plan. After a review of the technical information available and with the concurrence of the vendor representative the decision was made to transfer control to the stable "B" Regulator and disable the "A" Regulator by removing the "A" Steam Line Resonance Compensator (SLRC) card (A-42 card) from the electronic portion (see attachment) of the MT EHC Pressure Regulating System. This course of action would; 1) provide a stable and reliable pressure regulator, 2) eliminate the need for bias pot adjustments and 3) eliminate the potential loss of Reactor Pressure Vessel (RPV) pressure control if degradation of the "A" Regulator continued or accelerated and 4) utilize the least complex method of disabling the "A" Regulator. Similar card removals had been performed at Limerick Generating Station and
everal plants of other utilities without experiencing system transients.

The intent was to operate on the "B" Regulator only and leave the "A" Regulator out of service until the next shutdown.

At 2231 on 7/21/89 with Unit 2 at 79% thermal power, an attempt was made to remove the malfunctioning "A" Pressure Regulator Set from the electronic portion of the MT EHC Pressure Regulating System. This was to be accomplished by pulling the SLRC circuit card. The individual removing the card had difficulty on the first attempt, while repositioning his grip for a second attempt a slight side to side movement occurred. Before card removal, could be completed several relay actuations were heard and the Plant Manager in conjunction with the vendor representative halted removal.

Immediately, the Main Turbine Bypass and Control Valves opened, causing main steam line pressure to decrease to approximately 480 psig. At 850 psig main steam line pressure, a Group I Isolation occurred causing the Main Steam Isolation Valves (MSIVs) to close. As a result, a full reactor scram signal occurred due to MSIV closure. Initially, RPV level increased due to swell as a result of RPV pressure decrease, then decreased due to shrink as RPV pressure increased following MSIV closure.

This resulted in a Primary Containment Isolation System (EIIS:JM) Group II and III isolation signal as level decreased below 0 inches (172 inches above the core). Isolation of the Reactor Water Cleanup System (EIIS:CE), drywell equipment and floor drain sumps (EIIS:WD), drywell and torus instrument nitrogen supply (EIIS:LK), and reactor building ventilation (EIIS:VA) occurred as a result. RPV level decrease stopped at -35 inches, then level increased as Reactor Feed Pumps (EIIS:P) (RFP) "A", "B" and "C" continued to provide makeup. The RFPs were tripped before level exceeded +45 inches, level peaked at +75 inches and then decreased as automatic and manual Main Steam Relief Valve (MSRV) (EIIS:RV) operation reduced RPV inventory. Torus (EIIS:BS) cooling was placed in operation in anticipation of the decay heat removal requirements for the isolated reactor. RPV pressure increased to approximately 1100 psig during the transient and the following sequence occurred; 1) an Alternate Rod Insertion (ARI) High Pressure backup scram occurred at 2238 2) MRVs "J" and "H" lifted once automatically at 2242 and 3) the Operator manually cycled MSRVs "A", "B", "K" and "C" once each to reduce and maintain reactor pressure at 930 psig to 1060 psig between the time 2242 and 2258. Both Reactor Recirculation pumps tripped as expected during a 13 KV Auxiliary Bus (EIIS:BU) Fast Transfer caused by a MT trip initiated generator lockout. The Reactor Recirculation (EIIS:AD) pumps were subsequently restarted to provide forced circulation. The scram and Group II and III isolations signals were reset at 2305. The High Pressure Coolant Injection (EIIS:BJ) (HPCI) and Reactor Core Isolation Cooling (EIIS:BN) (RCIC) systems were placed in operation to control RPV pressure and level. HPCI was maintained in the full flow test mode (Condensate Storage (EIIS:KA) Tank (CST) to CST) to remove decay heat while RCIC Was utilized to control RPV level. The unit was stabilized in the Hot Shutdown condition at 2313.

Cause of the Event

The root cause of this event was the unusual service condition resulting from a malfunctioning component preceding the attempt to remove the malfunctioning Pressure Regulator Set from the electronic portion of the RPV Pressure Regulator System. The malfunctioning component was found to be the "A" Main Steam Pressure Transducer (PT-2184) (EIIS:PT) which drives the "A" Main Steam Pressure Sensor (Dwg. No. 832E840GR1) (EIIS:PE) which exhibited a consistent linear drift toward a lower Main Steam Pressure (i.e., more open control valves). Warm moist air from the moisture separator area condensed in the electrical conduit resulting in water damage to the transducer. The water damaged transducer affected the performance of the "A" pressure channel. This anomaly precipitated the evaluation, decisions and actions taken as addressed in the

Description of Event.

The cause of the EHC transient was determined to be the intermittent removal of the - 30 volt supply to the "A" Pressure Regulator and the voltage transients that resulted. This voltage fluctuation apparently was caused by the slight side to side movement that occurred during the attempt to remove the SLRC circuit card.

Analysis of the Event

No safety consequences occurred as a result of this event.

The Reactor Protection System (RPS) and Primary Containment Isolation Systems (PCIS) operated properly throughout the transient. The other Safety systems responded properly as described in the Description of Event. Due to the close proximity to full power at the start of the event, the expected plant response at 100% power would not have differed significantly.

Corrective Actions

The following corrective actions have been taken:

1. The Main Steam Pressure Sensor was returned to the vendor (GE) for analysis. It was evaluated and found not to have any defects.

2. The "A" Pressure Regulator circuit Cards with the exception of the SLRC and Operational Amplifier were replaced and calibrated.

3. The "A" SLRC has been bench tested satisfactorily without identifying any deficiencies.

4. RPV pressure has been monitored from the Control Room since returning to power and set pressure has not displayed any drifting characteristics.

5. This event has been reviewed by appropriate plant personnel.

6. The "A" and "B" Main Steam Pressure Transducers were sealed at the transducer to prevent water intrusion.

7. The Preventive Maintenance (PM) Program was enhanced to include: a) an inspection of each pressure transducer during each recalibration, and b) replacement of both pressure transducers every third

refueling outage.

The following corrective action is planned:

1. The conduit from the moisture separator area will also be sealed.

Previous Similar Events

There was one previous similar LER associated with troubleshooting the EHC System that resulted in a reactor scram. The following is a brief description of the event and those actions taken to prevent recurrence:

LER 3-87-02 addresses a low reactor water level scram caused by a main steam transient while performing testing on the EHC System. This testing was being performed to ensure the corrective actions taken in LER 3-87-01 to eliminate EHC signal oscillations were effective. Corrective actions for this LER included having a vendor representative and other personnel knowledgeable in the EHC System present while testing.

This action to prevent recurrence would not have prevented this event. Personnel experienced in the operation, maintenance and testing of the EHC System, including the Plant Manager and the knowledgeable vendor representative were present or involved throughout the testing/troubleshooting and decision making process.

TEXT PAGE 6 OF 6

Figure "Attachment 1, RPV Pressure Control" omitted.

ATTACHMENT 1 TO 9001250197 PAGE 1 OF 1

CCN-90-14012

PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION
R.D. 1, Box 208
Delta, Pennsylvania 17314
(717) 456-7014

PEACH BOTTOM-THE POWER OF EXCELLENCE

D.M. Smith
Vice President
January 16, 1990

Docket No. 50-277

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Unit 2

This LER involves a reactor scram resulting from an attempt to remove a malfunctioning Electro-Hydraulic Control System component from service. This revision provides additional information on the malfunctioning component.

Reference: Docket No. 50-277
Report Number: 2-89-015
Revision Number: 01
Event Date: 07/21/89
Report Date: 1/16/90
Facility: Peach Bottom Atomic Power Station
RD 1, Box 208, Delta, PA 17314

The revision to this LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv).

Sincerely,

cc: J.J. Lyash, USNRC Senior Resident Inspector
W.T. Russell, USNRC, Region I

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